4/p

Nuclear Materials & Propulsion Operation

INTRODUCTION TO NUCLEAR PROPULSION

Lecture 16 - NUCLEAR SAFETY

Authors - R. B. O'Brien
G. Briscoe
G. E. Denning
J. H. Lofthouse

Lecturer - C. L. Storrs

NASA CR 52946

FLIGHT PROPULSION LABORATORY DEPARTMENT
GENERAL ELECTRIC

SPT-10563 k

INTRODUCTION TO NUCLEAR PROPULSION

Lecture 16 - NUCLEAR SAFETY

Authors - R. B. O'Brien,
G. Briscoe,
G. E. Denning,
J. H. Lofthouse

Lecturer - C. L. Storrs

April 23, 24 and 25, 1963

Prepared for the George C. Marshall
Space Flight Center of the National
Aeronautics and Space Administration

Contract No. NAS8-5215

[redacted]

CONTENTS

CONTENTS (cont.)

1.0 INTRODUCTION

As was mentioned in the previous lecture, nuclear safety plays a major role in the operation of nuclear power plants. Among the reasons for this are the following: All the fuel for an extended period of operation is loaded into the reactor at the start of operation. Thus a large amount of emergy is potentially available which can conceivably lead to the release of radioactive fission products. A nuclear accident might thus have serious consequences beyond the test site itself. Furthermore, the effects of radiation on the human body are not fully understood or universally agreed upon. Finally, the reaction of the public has been a matter of much concern. For these reasons, a highly conservative safety philosophy has been universally applied to all reactor operations in this country.

2.0 GENERAL REQUIREMENTS OF NUCLEAR SAFETY CONTROL PROGRAM

The basic objective of a nuclear safety control program is the protection of personnel, plant facilities, and the surrounding community from the hazards of radiation and contamination which potentially could result from a nuclear incident or from the normal operation of a nuclear reactor.

2.1 Documentation and Approval

The following steps are typical of the efforts required to gain approval to build and operate a fixed-location, AEC-owned reactor and to provide the operator with internal procedural controls sufficient to assure the fulfillment of his obligations with respect to safe operation of such a nuclear system.

1. Pre-Operational Safety Analysis (Hazards Summary)

This document should convey sufficient information to allow a reasonable conclusion that the planned reactor can be built and operated safely in the proposed location. In general it contains a description of the reactor and its components, the site, facilities to be used, and an evaluation of the radiological hazards which could arise from postulated accidents. Appendix A contains an outline of the information ordinarily included. This document must be submitted to the cognizant AEC Operations Office and must receive their approval as well as that of the AEC Division of Reactor Development prior to operation of the powerplant. In many cases the AEC Division of Licensing and Regulation and the Advisory Committee on Reactor Safeguards will also review and approve the proposed operation. After approval has been obtained, the Hazards Summary is supposed to become of historical significance only.

2. Technical Operating Limits of Technical Specifications

Design and operating limitations that have an appreciable effect on the safety of reactor operation are compiled in a separate document. Most of these items were originally included in the final hazards summary. Appendix B outlines the detailed contents required. This document must also be reviewed and approved by the AEC prior to operation and is maintained as a current statement of pertinent limitations.

3. Standard Operating and Maintenance Procedures

Standard operating procedures are prepared containing explicit instructions for start-up and operation of the reactor with detailed checkout and calibration information provided. Maintenance to be performed is described in detail with special emphasis on limitations and control procedures to be observed. Responsibilities for individual operations and overall control are clearly delineated. While AEC approval of these procedures is not usually required, they are submitted for informational purposes.

4. Health Physics, Industrial Safety Procedures

Personnel protection standards for the control of radioactive material and radiation hazards are established from international policies of radiation protection, AEC requirements for control of radioactive materials, and working limits which have been adopted over a period of time at various AEC contractor sites throughout the United States. Normal industrial safety and industrial hygiene standards are maintained in addition to safety considerations of a nuclear nature. AEC approval is not required for these procedures.

5. Fissile Material Safety Procedures

Special procedures must be written covering specific storage arrangements and methods required in handling fissile materials. Each specific operation and particular form of material is examined and proper procedures determined. Because of the potentially serious consequences which could result from a criticality accident, the individual workers are not permitted any discretion in deviation from explicit handling and storage rules. AEC approval is not normally required.

6. Internal Safety Review

Prior to operation and periodically thereafter, internal safety reviews should be performed encompassing sufficient detailed evidence to permit the affirmative conviction that the reactor as built can and will be operated safely. The review should include examination of the start-up program, quality control or proof testing of manufactured items such as fuel and control rods, and a review of construction experience and system checkouts. The review is normally an internal matter for the operator and results may or may not be transmitted to the AEC at the discretion of the operator.

7. Permission to Operate

Upon satisfactory completion and review of the first two documents listed above, the AEC will grant permission to operate. The operating company's management should then provide the persons responsible for reactor operation with a formal letter granting permission to operate based upon the AEC approval and the availability of the other written procedures and a favorable internal review. Reactor operations then can be carried out, subject to the operational controls which have been developed.

2.2 Operational Control

The following internal controls are typical of those utilized to insure maintenance of safe operating conditions for a direct, open cycle nuclear system such as a nuclear rocket. From such a system, release of fission products to the atmosphere may result from normal reactor operation or from an unplanned release of radioactivity resulting from a nuclear accident. The seriousness of the resultant hazard, in either case, is strongly effected by the prevailing atmospheric diffusion conditions.

1. Evaluation of Environmental Conditions

In order to evaluate the hazards of radioactive effluent, it is necessary to possess rather complete information concerning the demographic and biological characteristics of potential receptor areas. A thorough study of these characteristics should be made prior to reactor operation and kept current.

2. Pre-Analysis of Controlling Accidents

For a given reactor a controlling accident may be selected for each phase of the testing program by weighing the assumed probability of occurrence and the magnitude of the hazard which would be created. Prior to reactor operation on any given day, the probable receptor areas for effluent hazards may be determined on the basis of current forecast weather conditions. A series of calculations can then be made of the significant doses to the various critical organs at occupied areas downwind resulting from fission product release should this controlling accident occur. If the controlling dose, which is dependent on atmospheric diffusion conditions, is calculated to be higher than acceptable permissible limits, reactor operation may be delayed until atmospheric diffusion conditions improve.

3. Pre-Analysis of Normal Operating Hazards

Prior to operation of the reactor, significant doses to the various critical organs may be calculated, per unit of reactor operating time, based on the expected continuous release of fission products, if any, during normal operating conditions. If the operating time is long enough, this dose calculation may be verified by measurements made in the field. The accumulation of dose in any defined receptor area, may be controlled by selecting desirable meteorological operating conditions and by controlling reactor operating time.

4. Decision to Operate

The decision to operate the reactor is based on a comparison of the calculated doses for all types of releases with the applicable limits for the particular test in question. Completion of Items 1, 2, and 3 above provides sufficient information to make this decision.

5. Measurement of Effects

During and following operation of the reactor, the expected fission product release rate may be compared with the release rate measured at the release point and at downwind field locations.

6. Post-Operational Analysis

Records maintained from reactor operation and existing meteorological conditions existing during the test period form the basis for analysis of the relationships of dose rate and weather conditions. Such analyses aid in upgrading the quality of dose rate predictions which are made.

7. Upgrading of Techniques and Procedures

Continuous effort should be maintained to develop and extend prediction and measurement techniques to arrive at the most realistic control system and most accurate dose evaluation methods consistant with practicality.

3.0 REQUIREMENTS OF SAFETY, TECHNICAL SPECIFICATIONS AND OTHER DOCUMENTATION

Specific details which must be included in documents submitted to the AEC and suggested schedules for their submission are outlined in this section and in Appendices A and B.

3.1 General Provisions of AEC Manual Chapter 8401

Requirements for submission and review of hazard report documents on AEC-owned reactors (excluding those covered by 10 CFR, Part 115)* are currently reflected in AEC Manual Chapter 8401. The following excerpts from this chapter are considered to be pertinent.

8401-01 Policy

"A hazard summary report shall be submitted prior to the commencement of construction of a new reactor, initial operation of a reactor, and significant modification in design or operating condition in a reactor under construction or in operation."

*Title 10 Code of Federal Regulations. Part 115 "Procedures for review of certain nuclear reactors exempted from licensing requirements".

8401-031 General

"A hazard summary report shall be submitted prior to the commencement of construction of a new reactor, initial operation of a reactor, and significant modification in design or operating condition in a reactor under construction or in operation"

8401-04 Definitions

"For the purposes of this directive, reactors include all apparatus, other than atomic weapons, designed or used to sustain nuclear fission in a self-supporting chain reaction including power, research, test, and production reactors, reactor experiments and critical assemblies."

"A significant modification of design or operating conditions, as used in the context of this manual chapter, is any modification which results in a substantial change in the existing safety characteristics of the reactor, and which does not clearly result in an improvement in the safety of the system."

Standards for content of hazards reports for AEC-owned reactors have not been formally issued; however, in view of the above policy statement, requirements established for hazards reports for licensed reactors and reflected in 10 CFR, Part 50.34 should be considered to define minimum standards. Since the technical specifications discussed below, are intended to establish safety limitations, the hazard report should be looked upon as a document supplying sufficient information for a meaningful safety appraisal of the proposed action. Significant background design and performance studies should be summarized and referenced in the report. The reviewer is interested not only in the adequacy of the design and the basis therefor, but also in the adequacy of the organization and proposed operational approach.

3.2 Typical Sequence in Processing Safety Analysis Submission to AEC

It is anticipated that the following sequence of actions will occur in the processing of a hazards report submission. The AEC operations office will vary according to contractual responsibilities involved.

1. Preparation and internal review by contractor organization prior to the development of an acceptable draft. (Participation by an AEC observer in this internal review is encouraged to assure familiarization with the proposed activity and for preliminary screening of questions, thus permitting more rapid handling of formal submission.)

2. Submission of draft hazards report to cognizant AEC operations office. (The operations office will normally provide copies to the Division of Reactor Development (DRD) for their information and preliminary comments.)

3. A meeting will be scheduled with the contractor to develop answers to operations office and DRD questions.

4. Following this meeting, comments on the draft will be provided to the contractor usually in written form, on any items unresolved in the discussion.

5. Formal submission to the operations office of the requested number of copies of the hazards report from the contractor.

6. The operations office will then prepare a written evaluation of the report and transmit this evaluation and the report formally to AEC Headquarters. (When reviewed by the Advisory Committee on Reactor Safeguards (ACRS) is contemplated, approximately 60 copies should be available although not all of these may be required. This will normally be for reports on site approval, construction approval, and approval for initial operation of new facilities. Where only DRD and DL&R review is anticipated, approximately fifteen copies should be available. This will normally involve all other types of submission.)

7. Review by the Department of Licensing and Regulation (DL&R), on an advisory basis to DRD for hazards reports submitted to Headquarters, is currently obligatory. Accordingly, a meeting with operations office, DRD, and DL&R personnel should be anticipated. (DL&R may in turn use the advisory capabilities of ACRS on major actions.)

8. DL&R will then formally advise DRD of their approval and any exceptions taken.

9. DRD will forward the results of the previous reviews through the appropriate assistant General Manager for final approval.

10. The operations office will then be formally advised that it may issue authorization to proceed to the contractor together with any exceptions or any requirements developed during the course of the reviews. (The operations office has the responsibility for seeing that these, and any additional requirements which it may desire, are in fact met.)

11. A formal authorization to proceed will then be issued by the operations office to the contractor.

3.3 Suggested Outline for Safety Analysis Report

The safety analysis report should follow the major topics listed below. A complete outline of information to be included in the safety analysis report is contained in Appendix A.

1. Summary
2. Introduction
3. Site or Environmental Analysis
4. Facility Description
5. Accident Analysis
6. Operational Procedures
7. Hazards Analysis
8. Evaluation of Facility's Hazards

3.4 Contents of Technical Specifications

A complete listing of technical specifications as currently required is reproduced in Appendix B.

Technical specifications are submitted as a separate document from the hazards report; however, they are submitted at the same time and essentially in the same manner as the hazards report. Some considerations to be used in writing technical specifications are as follows:

1. All matter included in and referenced by the technical specifications must be considered with extreme care with regard to:

   a. Distinguishing clearly between design parameters and operating limits.
   b. If specifications apply only to certain modes of operation - so stating.
   c. Stating clearly any exceptions such as "continuous operation not required," and "additional penetrations, conforming to ... may be made."
   d. Not including "information only" type statements.
   e. Considering time and effort consumed in obtaining a change in the specifications.

2. Tests to be applied to each item:

   a. Is it required by the letter and intent of AEC regulations?
      - If so, then the item should be included.
      - If not, is the safety importance so great that it should be included anyway? (Consider knowledge and ability of persons operating the plant five years from now.)
   b. Is the limiting number being quoted realistically? That is, can such accuracy of control, or such a flow rate, temperature, pressure, etc. be achieved? Is the highest (or lowest) value commensurate with safety?
   c. Is the piece of equipment so important that it should only be replaced with an exact duplicate?
      - If so, describe in necessary detail and include drawings if words do not suffice.
      - If not, describe functionally in terms of limiting design parameters so that any suitable piece of equipment may be substituted.
   d. Is the information to be included compatible with that given in the hazards report? In particular, is the limit compatible with the accident case analyzed in the report. A more extreme condition than that covered by analysis would not be appropriate.
   e. Can the information be used by an inspector to determine compliance with the license or operating approval. If not, provide backup reference material. (i.e., how measured or how correlated with measureable data.)

3.5 Other Documents Submitted to AEC for Information Only

Documents listed as Items 3,4,5, and 6 of Section 2.1 are submitted to the AEC for information purposes only. The information contained in these reports is of considerable use in evaluating the overall preparedness of the program.

3.6 Significant Changes

Changes which would invalidate the conclusions in the Hazards Summary report, all changes in the technical specifications, and changes to the operating procedures of a "new or unusual" nature and considered to be significant must be submitted for DRD approval prior to becoming effective.

4.0 POSTULATION OF ACCIDENTS

During the safety evaluation of a particular reactor all foreseeable accidents must be considered. The complete reactor system is studied and accidents are postulated assuming failure of various systems. Normally, in a well designed reactor system, independent multiple failures of components or systems must occur to produce an accident with serious hazards consequences. Some of the accidents postulated require so many independent failures that their occurrence is incredible.

4.1 Credibility

Webster's dictionary defines credible as "capable of being credited or worthy of belief". Some experts express credibility in terms of numerical probability of occurrence, while others feel that numerical estimates of a quantity so vague and uncertain have no meaning. The difficulty involved in assigning meaningful probabilities to the occurrence of a particular accident can result in considerable differences in opinion as to whether or not an accident is credible. The final decision is primarily a matter of judgment based partly on prior reactor design and operating experience and partly upon past precedent.

4.2 Maximum Credible Accident

The maximum credible accident (MCA) is that major accident (hypothesized for purposes of site analysis or postulated from considerations of possible accidental events), which would result in potential hazards not exceeded by those from any other accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

Application (Site Acceptability) - The question of suitability of a site for a reactor requires consideration not only of the factors influencing the factors influencing the probability of occurrence of an accident, but also the risk in terms of possible exposure of people to the hazardous consequences of such an accident.* The MCA is used as an aid in evaluating hazards involved at a particular site. In this evaluation the assumed fission product release from the core, the expected demonstrable leak rate from the containment, if any, and the meteorological conditions pertinent to the site are used to derive an exclusion area, a low population zone, and population center distance.

*See Title 10, Code of Federal Regulations, Part 100 "Reactor Site Criteria".

The exclusion area defined should be of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure. The location should have a low population zone of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

The population center distance should be at least 1 1/3 times the distance from the reactor to the outer boundary of the low population zone. In applying this guide due consideration should be given to the population distribution within the population center. Where very large cities are involved, a greater distance may be necessary because of total integrated population dose considerations.

4.3 Other Controlling Accidents

The Maximum Credible Accident may be incredible at some stage of operations. For example, if the MCA is a start-up accident, then it would not be a credible event during the course of normal reactor operations at power. In addition, the MCA may be felt to have such a low probability of occurrence that other, more reasonable events should be considered as well. For times when the MCA does not apply, another controlling accident should be defined. While this would be of lesser consequence than the MCA, it would be considered as a major factor in determining whether or not the reactor is to be operated at a given time or for a specific test operation.

4.4 Typical Accident Initiating Events

1. Set-Up Mechanisms

During the preparation of a hazards summary a systematic review of all power plant components is performed to determine the consequences resulting from the failure of any parts or system of the power plant. Since no single part or system may be expected to function without failure at all times, the power plant must be so designed that no single event or sequence of events initiated by a single event will result in unacceptable hazards. Some of the various mechanisms by which a nuclear powerplant might be "set-up" for subsequent accident are as follows:

Failure of Safety Systems - The safety systems include sensors which measure parameters such as temperature, reactivity, pressure, and radiation levels. These sensors frequently activate devices for alerting the operator and/or shutting down the reactor when unsafe conditions arise. Failure of a single safety system, in itself, should not result in reactor damage. To postulate many of the accidents analyzed, such as unrestrained control rod withdrawal, it is necessary to assume multiple failures of safety systems.

Failure of Control Systems - Control system failure which could result in damage to the reactor might result from saturation of sensors which transmit false low indications of reactor power levels. Based on this signal, the servo mechanism or the operator then continues to pull control rods until the power level is high enough to melt portions of the core.

Since it is impossible to determine all of the events which could result in failure of the control system, the usual procedure is to assume failure of various control systems with simultaneous failure of the safety system and to analyze the consequences.

Procedure Error - In the preparation of procedures it is possible to overlook some step or interlock which could increase the safety of the system. Evaluation of operating procedures by a hazards engineer independent of operating supervision helps to reduce this possibility.

Operator Error - Failure of the operator to follow procedures can result in power plant damage. Such an error could result from the omission of a step in pre-startup checkout procedure or inattention on the part of the operator. The possibility of such errors is reduced by stringent qualification requirements for reactor operators. In addition, most errors committed by an operator would require simultaneous safety system failure to result in reactor damage.

Sabotage - Deliberately blowing up the power plant at a critical time could result in release of fission products to the atmosphere. Normal security measures and the large amount of explosives required make this type of accident extremely unlikely. A more probable type of sabotage would be deliberately causing malfunction of one or more of the reactor safety systems, thus setting the system up for an accident during the next operating period.

2. Reactivity Induced Accidents

Reactivity accidents are defined as those accidents that result from the addition of more excess reactivity to the reactor core than is required for normal changes in power level. This results in an increase in power level which can, if allowed to continue, result in physical damage to the core and subsequent release of fission products to the coolant.

Unrestrained Control Element Action - To analyze an accident caused by unrestrained control element action, it is normal practice to assume that all such elements that may be operated simultaneously are moved in the direction of increasing reactivity at the maximum rate physically possible. All normal safety systems are assumed to be inoperative so that the excursion is limited only by the various reactivity decreasing mechanisms inherent in the design.

Flooding - Flooding of gas cooled reactor with a hydrogenous fluid is always a possibility. If the probability is considered great enough, it may be required that control rod worth be sufficient to insure a subcritical flooded assembly. If flooding is considered a credible event for the particular system, the addition of reactivity by this means should be considered as a possible accident initiating mechanism.

Geometry Changes - Based on the design of the reactor, various changes may result in either increased or decreased reactivity. Possible mechanisms for changing reactor geometry could be temperature or pressure changes, explosion, implosion, mechanical failure, missiles, etc. Potential initiating mechanisms should be analyzed to insure that optimum safety has been achieved.

Temperature Reactivity Effects - The magnitude of potential power excursions may be considerably reduced if negative temperature reactivity coefficients are inherent in the design. In many water moderated designs, the fact that the water becomes less dense and hence decreases in moderating value as the temperature increases has an important effect. In other designs, such effects may be unimportant or may be positive. Formation of steam bubbles in the water within reactor has the same effect of reducing the density of moderator water. Expansion of other materials within the core may also have a significant effect upon reactivity. The negative Doppler coefficient reduces reactivity following an increase in fuel temperature as a result of increased resonance absorption of neutrons, frequently in U^{238}. The relative value of temperature coefficients depends on various parameters such as the rate of power rise. For instance, the fuel temperature coefficient may be of much greater importance than other effects for a short period excursion because the fuel temperature rises much more rapidly with flux than do the coolant or moderator temperatures. All of these possibilities must be considered in evaluating the power generated by a reactivity induced incident.

Other - Special types of reactors may have mechanisms for introducing unwanted positive reactivity. Each reactor must be considered separately to cover all possible causes of reactivity induced accidents.

3. Loss of Cooling Accidents

Loss of coolant accidents are those in which the ability to remove the heat generated in the reactor fuel is lost. Melting of a portion of the core generally results unless the system can be shut down and adequate aftercooling provided before such damage occurs.

Flow Maldistribution - Introduction of foreign objects into the flow stream could result in reduced flow and subsequent localized melting of the reactor core.

Loss of Coolant Flow - Loss of cooling may be caused by failure of the blower, pump or other mechanism for forcing coolant through the reactor. Mechanical failure in the piping of the cooling system could also result in a complete loss of coolant from the reactor.

Loss of Aftercooling - Even after a reactor is shut down decay power from fission products may be great enough to melt a portion of the fuel unless some coolant flow is maintained for a reasonable time after reactor shutdown.

4. Mechanically or Chemically Induced

Abrasion, Erosion - Excessive erosion or abrasion of fuel element cladding can result in loss of fission products to the coolant stream. The magnitude of the release and subsequent effects must be estimated unless data from environmental samples are available.

Oxidation - Combustion - Hazards of combustible coolant, moderator or structural materials used in specialized reactors must be considered. Oxidation of internal reactor components, especially fuel cladding, may seriously reduce integrity with a resultant release of fission products.

Missiles - "Missiles" consisting of pieces broken loose from rotating parts such as turbine blades could conceivably puncture piping or vessel walls with a resultant loss of cooling and/or damage to vital components.

Chemical Explosion - Reactions may occur between structural materials and coolants at elevated temperatures during an accident or even during normal operation, and could result in an explosion. Possible accidents of this type must be considered.

4.5 Typical Accident Terminating or Limiting Mechanisms

A reactor excursion will result in effects that will terminate the chain reaction and limit the power generated in the excursion. The total power generated will depend on the rate of power rise (the amount of excess reactivity) and the time lag between the power rise and time required for the inherent shutdown mechanisms to remove sufficient reactivity to reverse the excursion.

1. Meltdown - During a power excursion the heat generated in the fuel rises faster than the heat can be transferred to the cooling medium. As the fuel becomes molten, the geometry of the core will be altered, and the gravitational or forced flow of the molten fuel will reduce the reactivity and shut the reactor down.

2. Vaporization - If the power rise is rapid, the rearrangement of the molten fuel may not be rapid enough to shut the reactor down before vaporization occurs. In this case a rapid expulsion of fuel vapor will terminate the excursion.

3. Core Expansion - Steam or vapor pressure may produce an outward movement of fuel elements and structural components as occurred in the SL-1 accident. This will shut the reactor down if the reactor design will permit a large enough change in geometry.

4. Temperature Reactivity Effects - If the temperature reactivity effect is large enough, it may serve to terminate the excursion, possibly without damage to the reactor.

5. Ejection of Fuel (Mechanical Failure) - Temperature and/or pressure stresses may cause failure of the fuel latching mechanisms or support structure with subsequent ejection of fuel elements or portions thereof from the core, thus terminating the reaction.

4.6 Fission Product Release

Since the release of large quantities of fission products to the atmosphere is highly undesirable due to man's low tolerance of radioactive materials, as many barriers as practical should be included in the reactor design. For instance, fission products in a typical commercial power reactor must penetrate four separate barriers prior to release to the atmosphere. These are: (1) fuel matrix, (2) fuel element cladding, (3) reactor vessel and (4) containment vessel. The result is that, even under the maximum credible accident, only a small part, if any, of the fission products are released to the atmosphere. An open cycle system, however, normally has only the first two of these barriers to fission product release.

1. From Molten Fuel

The percentage of fission products released from molten fuel depends on a number of parameters.

The melting temperature will be determined by the chemical composition of the fuel. Uranium metal has a melting point of 1133°C while the melting point of UO_2 is 2176°C.

Very little of the fission products which remain solid will be released from molten fuel while a much greater quantity of those which volatilize will be released. Since fission products constitute a large number of elements with a wide spectrum of boiling points, the melting temperature of the fuel will determine to a large extent the percentage of volatile fission products and, consequently, the overall release percentage. Since vaporization procedes at a finite rate the time that the fuel remains above fusion temperatures of the various fission product elements or compounds will also have an effect on the per cent release.

2. From Unmelted Fuel

The amount of fission products released from the unmelted portion of the fuel is usually negligible compared to the overall release during a reactor excursion involving core meltdown or vaporization. Significant quantities of the noble gases and halogens may be released by diffusion if fuel temperatures remain elevated, but below the melting point for significant time periods, however.

3. From Vaporized Fuel

All of the fission products are conservatively assumed to be released from volatilized fuel. Subsequent release to the environment depends on factors discussed below.

4. Release to Environment

The amount of fission products which are subsequently released to the environment after release from fuel elements in the reactor core may be reduced by a number of factors.

The Containment - Closed cycle nuclear power plants usually are provided with a containment system; a large vessel completely encompassing the pressure vessel and reactor structure. This vessel is designed to withstand pressures created by the maximum credible accident and ideally prevents the release of fission products to the environment. No practical completely leak proof system has yet been devised. However, a small release rate (2%/day) from the containment vessel is usually acceptable.

Coolant Entrainment - Any coolant lost from the reactor core or pressure vessel to the containment vessel or atmosphere will carry with it a large fraction of the fission products previously released to the coolant.

Plateout on Cool Surfaces - Many of the fission products will adsorb on/or combine chemically with structural surfaces with which they come in contact. The halogens plate-out readily on almost any cool surface. A part of the volatilized fission products are precipated or plated out as temperatures drop during passage out of the reactor. Experiments indicate that the percentage plated out can vary widely depending on specific conditions such as temperature and passage time. A typical assumption for a specific system might be that 0% of Noble Gases, 50% of halogens, and 70% of others plate out on cool surfaces and thus, are not released to the environment. In some systems, "rainout" due to condensation of vaporized coolant may remove larger fractions.

Effluent Cleaning - Filters and/or other cleaning media for normal effluent will further reduce the amount of radioactive material released since a large part of the effluent resulting from an accident will probably be exhausted through building or facility exhaust filters.

5. Typical Release Estimates

A number of release experiments have been conducted for specific reactor types and more are currently planned by the AEC. These experiments have been necessarily conducted for specific types of fuel and cladding, at specific temperatures and under specialized conditions. The data is not yet complete enough to be of general use in calculating accidental release of fission products.

The former ANP practice was to postulate an assumed release fraction, and then verify the release experimentally. Accordingly several experiments were conducted where coolant flow to one fuel element was deliberately blocked and the element was allowed to melt with a subsequent release of fission products to the environment.

These experiments were carried out under meteorological control with a radiation monitoring grid downwind established to correlate radioactivity concentrations downwind with release rates measured by the stack monitoring system. In the summer of 1959 the GE Hazards Council met in Gatlinburg, Tennessee, and formulated a standarized format for estimating fission products released from reactor accidents if more specific data was not available. This is reproduced in Table I. The specific numerical valves of Table I are selected as representative of conditions that might be expected following an accident in a reactor of the boiling or pressurized water type, but they may also be in the right range for some other systems. Modifications can be made, of course, for other assumptions or designs.

Table I

WEIGHTED FISSION PRODUCT RELEASE

Class of f.p.	% f.p. in reactor fuel	% released from fuel	% not plated out on cool surface	weighted % released to the primary loop for 50% core melt
Noble Gases	10	100	100	5
Halogens	10	50	50	1.3
Volatile Solids	11	50	30	0.8
All Others	69	1	30	0.1
				7.2

5.0 NORMAL OPERATING HAZARDS

During normal reactor operations certain radiation hazards are present which must be considered. Direct radiation from the power plant normally presents problems which are solved by a combination of reactor and facility design and administrative control. Release of radioactive material to the environment is another problem which requires consideration. Pre-analysis of expected fission or activation product release during normal operations must be made and adequate safeguards provided to insure that any such release is kept within non hazardous limits.

5.1 Fission Product Release

Even a minute release of fission products from the fuel structure, at a continuous rate, may have serious consequences during normal operations. If fuel cladding integrity is maintained, loss of fission products whose volatization temperature is less than operating temperature can occur by gaseous diffusion through the matrix and cladding. When fuel cladding integrity is violated fission products may be ejected through the breach by direct recoil during the fission process.

Loss of cladding integrity could progress slowly by erosion, corrosion, or chemical action of minute quantities of foreign matter in the coolant stream. Vapor pressures within the fuel could create blisters at localized hot spots. In this case loss of fission products could occur from a combination of recoil and diffusion.

5.2 Direct Radiation

Radiation dose rates external to the power plant shielding during normal operation are a result of various sources of activity such as:

1. Direct radiation leakage from the reactor core through the shield
2. Neutron induced activity (such as A^{41} and N^{16}) in the moderator and/or coolant.
3. Fission product activity in the coolant.
4. Effluent activity in gaseous waste handling systems.

6.0 MAINTENANCE AND SERVICING HAZARDS

6.1 Reactor Accidents during Maintenance Operations

Mechanisms for initiating reactor accidents as discussed in section 4 apply to both operations and maintenance.

The probability that an accident will occur during servicing or maintenance operations is higher than during reactor operation for at least two reasons: (1) An increased number and variety of workers are involved, and (2) Servicing frequently involves unforeseen maintenance and repairs, and procedures for dealing with them are not as explicit as normal operating procedures.

The effects of an accident occurring during maintenance or servicing will probably be more severe than for other types of accidents for the following reasons: Men working directly on the power plant would receive larger direct radiation doses than would be likely under other circumstances. Since the pressure vessel and/or containment vessel would probably be breached during such operations, a consequent larger release of fission products to the environment could be expected.

6.2 Fuel Handling and Storage Accidents

Because of the hazards that could result from an accidental chain reaction, stringent requirements governing the handling and storage of reactor fuel have been established to prevent such an occurrence. Each individual case of fuel storage or handling is studied and acceptable conditions for performing the function are specified. All activities in the storage and handling of fuel are supervised by trained personnel as an additional safeguard.

1. Control Parameters

The amount of fissionable material necessary to cause a chain reaction during handling and storage depends on a large number of parameters. A few of the more important ones are:

Mass - A minimum of approximately 2 lbs. of uranium is necessary to sustain a chain reaction under moderated conditions.

Geometry - A spherical shaped mass is more reactive than any other.

Moderation - Neutrons must be moderated (slowed down) by some material such as water, before the fission chain reaction will occur with a small amount of U^{235}. Without moderation a minimum of about 45 lbs. of uranium is necessary to sustain a chain reaction.

Interaction - It is possible for two subcritical systems to be made critical by bringing them into close proximity. Other control parameters are reflection, density, homogeniety, fuel enrichment, and poisons.

2. Procedural Controls

In practice one or two parameters are controlled during handling and storage with others considered at optimum reactivity. In general, the mass is limited by container size and the individual containers are separated by physical barriers. Whenever possible handling operations are conducted under conditions where criticality is virtually impossible, and as little dependence as feasible is placed on routine personnel actions.

3. Potential Results

It is not expected that a nuclear accident resulting from improper storage or handling of fissionable material could result in an explosion compared to even the earliest of the atomic bombs. However, an accident of this kind could result in radiation levels that might be lethal. If such an accident generated sufficient heat to melt metals, extensive contamination of equipment and facilities could result. Radiation and contamination levels conceivably could require that many months elapse prior to the safe resumption of operations. The incident can probably be best compared to an accidental chemical reaction accompanied by the release of considerable quantities of toxic materials.

6.3 Direct Radiation

Workers must be protected from direct radiation during the transfer of irradiated reactor cores and fuel elements, while handling radioactive waste, and while working on contaminated or activated equipment. Accurate shielding calculations and carefully prepared procedures covering these operations are necessary to adequately safeguard personnel.

6.4 Airborne Activity

Airborne activity is especially hazardous in that many of the fission products tend to remain in the body after deposition in the lung. Sources of airborne activity include re-entrainment of surface contamination created during previous operation or accident.

While the source of fission gas activity is greatest during reactor operation, fission gases may continue to escape from the reactor after shutdown, creating hazards during servicing for a period of days.

6.5 Surface Contamination

Surface contamination is defined as radioactive materials deposited on the surface of objects. Contamination provides a direct radiation hazard, and this can be a health hazard if it comes in direct contact with the skin or is inhaled.

6.6 Radioactive Waste Disposal

If the concentration of radioactive material is above permissible limits for discharge to the environment, the radioactive material is either stored or packaged and buried.either in the earth or deep sea. Packaging and burial must be such that there is reasonable assurance that hazardous amounts of radioactive material will not be released to the environment at any time in the future.

7.0 RADIOLOGICAL HAZARDS - BIOLOGICAL CONSIDERATIONS

7.1 External Radiation

External radiation is that radiation, affecting a biological system, originating from sources outside the system.

1. Range and Penetration

The range and penetrating ability of the various types of radiation varies greatly. Alpha radiation has a range of only a few centimeters in air and will not penetrate the skin sufficiently to be considered an external hazard. The range of beta radiation may be as great as several yards in air but its penetrating ability is such that, if the eyes are shielded, the dose received from it is not considered a whole body dose, but only a skin dose.

Gamma and neutron radiation is absorbed exponentially and is considered a whole body dose.

2. Radiation Units

Roentgen (r)

Definition - The quantity of X or gamma radiation such that the associated corpuscular emission per 0.001293 grams of air produces, in air, ions carrying one electrostatic unit of quantity of electricity of either sign.

Application - The roentgen is the term used to express the amount of X or gamma radiation delivered to a specified area or to a part of or the whole body.

Roentgen Absorbed Dose (rad)

Definition - The unit of absorbed dose equivalent to 100 ergs/gram.

Application - The rad is the term used to express the amount of beta radiation, or the total of all beta plus gamma radiation absorbed in a specified area or to a part of or the whole body of any material. If the term is to be used to express an amount of beta radiation it should be written 1 rad, beta; if the term is to be used to express a mixture of beta and gamma radiation it should be written - 1 rad.

Roentgen Equivalent Man (rem)

Definition - That quantity of any type ionizing radiation which when abosrbed by man produces an effect equivalent to the absorption by man of one roentgen of X or gamma radiation (400 KEV).

Application - The rem is the term used to express the amount of all radiation including neutrons delivered to a specified area or to part of or the whole body. When the term is used to express neutron dose it should be written - 1 rem-neutrons; when used to express total dose including neutrons it should be written - 1 rem.

Curie (c)

A curie is that quantity of radioactive material that disintegrates at a rate of 3.7×10^{10} disintegrations per second.

A microcure (10^{-6} curies) is equivalent to 3.7×10^4 disintegrations per second.

A millicurie (10^{-3} curies) is equivalent to 3.7×10^7 disintegrations per second.

3. Relative Biological Effectiveness (RBE)

Relative Biological Effectiveness is the ratio of gamma or X-ray dose that is required to produce a given biological effect to the dose of a particular type of radiation which would cause the same effect.

7.2 Internal Exposure

Internal exposures is that radiation affecting a biological system originating from sources within the system.

1. Intake

The various ways that radioactive material can enter the body are:

Inhalation
Ingestion
Absorption through the skin
Skin openings - (Material may enter directly into eyes, ears, cuts or abrasions.)

2. Internal Depositions

The per cent of radioactive material entering the body which becomes deposited within the body depends on the method of intake and the form of radioactive material.

The penetration and retention of particles in the lungs is a function of particle size. Soluble material is more readily retained in the body than is insoluble material regardless of the method of entry.

Different chemical forms of the radioactive material tend to concentrate in various organs of the body. For instance, radioactive iodine is concentrated in the thyroid, while fission products related to calcium, such as Sr^{90}, are concentrated in the bone.

3. Critical Organ Concept

Since body organs are not all equally radio-sensitive nor equally vital to the well being of the entire body, it is not necessarily true that the dose to the organ accumulating the greatest concentration of radioactive material will result in the greatest over-all damage to the body.

The critical organ for a given isotope is therefore defined as that organ receiving the isotope that results in the greatest over-all damage to the body.

4. Units

Units of internal radiation exposures are the same as for external exposures.

7.3 Permissible Limits

1. Basis for Establishing Radiation Protection Guides

A dose below which no possibility of genetic or somatic damage exists has not been established for ionizing radiation. In the absence of an established threshold dose it is considered wise to avoid all unnecessary exposure to radionuclides. Accordingly a permissible radiation dose limit is not that does which suddenly becomes hazardous at that point. Therefore, doses should be kept at the lowest practical level. Also, radiation guides should not be accepted as absolute limits. Rather they should be regarded as that dose which should not be exceeded without careful consideration of both genetic and somatic effects.

Radiation induced deleterious mutations may be passed on to future generations. The total damage is dependent on the total integrated dose received by the population and is relatively independent of the number of individuals exposed.

Also, since the total number of radiation workers is small compared to the total population, it has been possible to set "occupational" radiation limits higher than those for the general public. These limits are generally based on the absence of observed effects at these low doses. However, it is expected that, in the light of present knowledge, occupational exposure for the working life of an individual at the recommended maximum permissible values is not expected to entail appreciable risk to the individual or to present a hazard more severe than those commonly accepted in other present-day industries.

2. Occupational Whole Body Radiation Dose Protection Guides

The guides established for occupational whole body radiation are:

Accumulated Lifetime total - 5(N-18) Rem

where N = the age of the individual in years

Annual - 12 Rem

Quarterly (any 13 week period) - 3 Rem

3. Occupational Critical Organ Dose Protection Guides

The Federal Radiation Council has recommended the following critical organ dose guides:

Skin and Thyroid	10 Rem/qtr; 30 Rem/yr.
Extremities	25 Rem/qtr; 75 Rem/yr.
Eyes, Gonads, Head, Trunk, and Blood-forming Organs	Same as whole body
Bone	Equivalent to 0.1 μg Ra^{226}
Others	5 Rem/qtr; 15 Rem/yr.

4. Population Dose Recommendations

The Federal Radiation Council has recommended limits for public exposures from sources other than that received for medical purposes and from natural sources as follows:

Whole Body

Individual - 0.5 Rem/yr.
Average - 5.0 Rem/30 yrs.
Sample of exposed population - 0.17 Rem/yr. per capita

Thyroid

Individual - 1.5 Rem/yr.
Sample of exposed population - 0.5 Rem/yr.

5. Accepted Emergency Dose Limits

The emergency dose limit is that dose which may be accepted to save lives or valuable property. It is limited to once in a lifetime of any individual and need not be included in the lifetime total dose formula: 5(N-18) Rem.

The occupational whole body emergency dose limit is 25 Rem. Although no specific recommendation has been established for the general population, a limit of 25 Rem is usually accepted and is frequently used in reactor siting criteria. No specific recommendation has been made for a thyroid emergency dose limit. A 300 Rem limit is usually accepted for both occupational and public exposures.

8.0 CALCULATION OF EFFLUENT HAZARDS

The calculation of effluent hazards involves the calculation of three factors: the source term representing the amount of radioactivity release to the environment, the biological term giving the dose received by the population, and the transport term describing the dispersion of the effluent.

8.1 The Source Term

The source term used in estimating effluent hazards is determined for either the accident case as outlined in Section 4 or for the normal operating case as in Section 5. For the accident case the source term may be quite closely defined by the conditions imposed in postulating the accident. If experimental data is available from fuel sample tests, in environments similar to those proposed for the normal operating conditions, the normal operating source term may be quite closely defined; however, when such data is not available, one must resort to "best estimates" to define a source term until such time as operating data become available. Often an effluent release will contain radioactive material in both the gaseous and particulate forms.

8.2 The Biological Term

The dose received by a population downwind from a radioactive release is the sum of doses received by several modes of exposure.

1. Exposure Modes

External - Cloud Passage - The external dose due to passage of the effluent cloud is that dose received from radiation emanating from the radioactive constituents of the cloud while they are airborne. A receptor on the ground may be exposed to the radiation from the cloud while the cloud is above him or he may be immersed in the cloud. Of prime importance is deep penetrating gamma radiation giving rise to "whole body" exposure; although technically speaking, the beta radiation dose (shallow penetration) to the skin, eyes, gonads, etc., may be considered if the receptor is immersed in the cloud during its passage.

Inhalation - Cloud Passage - Dose to Thyroid, Lung, and Bone - While the receptor is immersed in the effluent cloud during its passage he will inhale some of the radioactive material. A portion of this material, depending on such factors as particle size and breathing rates, will be retained in the respiratory passages, the smaller particles penetrating into the alveolar regions of the lung. Those materials of proper solubilities will subsequently enter the blood stream. Depending on the chemical properties of the various isotopes entering the blood stream various body organs may be affected. For the radioactive isotopes of iodine the thyroid becomes the critical organ. Isotopes of strontium, barium, promethium, etc., tend to collect in the bone. Once the radioactive material is deposited in a body organ, the beta disintegration energy for those isotopes is absorbed by the organ and is usually expressed in units of rad dose.

External - Ground Deposition - Fallout, Rainout, Plateout - Another source of radiation to a downwind receptor resulting from passage of an effluent cloud is that from the deposition of radioactive material on the ground or on vegetation near the ground creating a plane source of external radiation that continues to exist after cloud passage. This source is the result of three mechanisms of deposition; namely, fallout, rainout, and plateout. Fallout is simply the ground deposition of airborne particles under the force of gravitation. The rate of deposition is a function of the particle size distribution. Rainout is a result of the constituents of the effluent cloud being caught by rain droplets either through particle entrainment or adsorption of the gases by the droplet. Plateout of radioactivity results from the intimate contact of the gaseous material in the effluent cloud with surface areas. The gases "plateout" by absorption-adsorption processes.

Ground deposition results in depletion of the cloud and, consequently, reduces the dose to receptors further downwind.

Ingestion - Biological Cycles - Ingestion of radioactive material by man may occur by two important processes: (1) Elimination of particulates from the respiratory tract by cilliary movement into the throat where it is consequently swallowed. (2) Ingestion of food or water contaminated with radioactive material. As outlined in the previous paragraph, effluent clouds may contaminate soil and leafy vegetables resulting in incorporation of radioactive material into a biological cycle. Biological systems and biological cycles tend to concentrate radioactive material. Example: An effluent cloud containing radioactive isotopes of iodine contaminates a pasture land by plateout. Milk cows graze on the contaminated grass concentrating the iodine in the milk. The milk is consumed by man whose digestive system transfers the iodine to the thyroid. This is a rather efficient cycle resulting in the concentration of radioactivity in a small organ. The resulting dose to the thyroid may be orders of magnitude greater than the dose to any other organ from any of the other modes of exposure discussed so far. This becomes an especially important consideration if the milk is consumed by infants and small children whose diet consists mainly of milk and whose thyroid is small.

2. Biological Constants used in Dose Calculations

Standard Man - Recommended Values Available - ICRP - The calculation of dose to a biological system such as man requires some knowledge of the bio-physical processes involved. This is especially true of internal dose. Many of the constants used in dose calculations have been essentially standardized by use of a convention called "standard man" which defines such parameters as size, weight, density, elemental exchange rates, intake and output of the body or body organs. Recommended values of these parameters have been tabulated for use in dose calculations. These appear in National Bureau of Standards handbooks and in various other publications, journals, and handbooks. The most active group involved in standardizing calculational techniques and recommending standard values is the International Commission on Radiation Protection. Results of their work are published in the "Health Physics Journal".

Special Cases - Use of "standard Man" parameters when considering dose to a population from an effluent cloud passage is not always precisely applicable. Special or limiting cases arise for some situations. A case in point is the concentration of iodine isotopes by the biological cycle mentioned in 7.2.1.4. The mass of a child's thyroid is $\sim$1/10 that of a standard man; therefore, the dose to a child's thyroid will be $\sim$10 times greater than to a standard man for the same amount of radio-iodine ingested.

8.3 Transport and Dispersion of Effluent

Once the effluent release is beyond physical control of the plant, atmospheric conditions, and to some extent the condition of the effluent, will determine how the cloud or plume will be transported and dispersed downwind from the release point.

1. Initial Dilution - Turbulent Wake of Buildings - Stack Augmentation

As the release leaves the physical confines of the building, test facility, or stack, significant initial dilution may take place by air turbulence in the wind wake of the building. If the release is through a stack, augmentation of the stack draft may cause considerable initial dilution. Initial dilution is an important consideration especially if short diffusion distances are being considered (i.e., if the receptor is a short distance downwind).

2. Effective Release Height - Temperature Velocity, Volume, Wind Velocity

An important consideration in estimating down wind dose from effluent is the determination of an effective release height. Effective release height is the initial stack, chimney, or vent height plus the rise of the gas due to its temperature, velocity, etc. Several mathematical models have been proposed for estimating this effect and some have been partially verified for certain conditions. Effective release height becomes less important in the hazard calculation if the distance downwind to the point of interest is very much larger than the effective release height. Effective release heights for nuclear rocket tests may be great, depending on the orientation of the nozzle. Unless large diffusion distances are available to downwind populations, effective release height corrections to hazard calculations will become highly important.

3. Atmospheric Conditions

Atmospheric diffusion of an effluent is usually thought of in terms of a turbulent motion or mixing process. Parameters affecting this mixing process are wind speed, wind direction shifts, wind shear (change of wind direction with height) and atmospheric stability which is usually defined as a function of the temperature gradient with height.

Wind Speed - The overall effect of wind speed on cloud dispersion is to increase diffusion with an increase in velocity, thus reducing the concentration of the cloud and the resulting dose. Offsetting this beneficial effect, however, is the more rapid transport of the effluent allowing less time for radioactive decay before reaching the population of concern. This is an important consideration when dealing with postulated release of fresh fission products from an operating reactor.

Wind Direction - Wind direction fluctuations tend to disperse the cloud more rapidly. The effect on dispersion is a function of the magnitude, frequency, and shape of the fluctuation. High frequency shifts are important when considering an instantaneous or puff release. Added to this effect are the low frequency shifts (upon which high frequency fluctuations may be imposed) sometimes referred to as meander, which become more important when considering continuous release of effluent. An analysis of annual and seasonal wind direction data may be desirable when performing calculations for site selection or when designing an operational control program in which it may be desirable to utilize winds from a particular sector to disperse the effluent away from human populations. This data is usually presented in the form of a wind rose.

Wind Shear - Wind shear is the gradual change of wind direction with height and only becomes important when considering release with high effective release heights such as that from high effective stacks or rocket exhausts.

Atmospheric Stability - Temperature Gradients - The stability of the atmosphere is usually expressed as a function of the temperature gradient with height. When the atmospheric temperature decreases with height the condition is defined as a lapse (a normal day time condition). When temperature increases with height through a layer of atmosphere the condition is defined as an inversion (usually a nocturnal condition). A lapse condition characterizes an unstable atmosphere (good effluent diffusion). An inversion condition characterizes a stable atmosphere (poor effluent diffusion). The depth of inversion layers may vary widely, from a few hundred feet to several thousand feet. An intermediate or isothermal condition exists if there is no change in temperature with height through a layer of atmosphere.

Mechanical Turbulence - Mechanical turbulence as opposed to the thermal turbulence discussed above, is a function, primarily, of terrain roughness. Hence atmospheric diffusion problems rely on the ability to interpret these conditions in proper terms which can be expressed by a mathematical model.

4. Mathematical Models

A number of mathematical models have been developed to express atmospheric diffusion of air pollutants. Of these, the equations of Roberts (Ref.1), Bosanquet & Pearson (Ref.2), and Sutton (Ref.1) are possibly the best known. The equation of Sutton has been widely used at atomic energy installations to predict effluent diffusion. This equation is in some respects more flexible than other models and has met with general success in the verification process. The equation contains three constants for the completely non-isotropic point source. The values of the constants are functions of the wind velocity, terrain roughness and temperature distribution in the atmosphere. In general the equation has sufficient flexibility to be applicable to all meteorological conditions. To a receptor on the plumb centerline at ground level downwind from a continuous point source the cloud concentration is usually expressed as follows:

$$\chi = \frac{Q\, e^{-h^2/\sigma_z^2}}{\pi\, \bar{\mu}\, \sigma_y\, \sigma_z}$$

where χ = Concentration (curies/meter^3, grams/meter^3 etc.)
Q = Source strength (curies/sec, grams/sec etc.)
h = Effective stack height (meters)
$\bar{\mu}$ = Average wind speed (meters/sec)
σ_z, σ_y = Standard deviation of gaussian distribution of the cencentration in the vertical and lateral directions with respect to the wind direction coincident with the x axis.

$$\sigma_z = \frac{1}{\sqrt{2}} c_z\, \chi^{(2-n)/2}$$

$$\sigma_y = \frac{1}{\sqrt{2}} c_y \chi^{(2-n)/2}$$

n = Atmospheric stability parameter (dimensionless)
χ = Distance downwind (meters)
c = Diffusion coefficient (meters n/2)

The factor 2 is included to reflect the assumption that the ground is a reflecting plane thus doubling the concentration to a receptor at ground level. The diffusion coefficient c is also a function of the release height, that is, turbulence and mixing tends to decrease with height. This correlation has not been well defined for great effective stack heights such as those that may be expected for certain rocket firings or high effective stacks.

9.0 SUMMARY

Safety and hazards analysis requires many engineering disciplines to be understood and coordinated in order to effect a truly satisfactory solution. In general it may be stated that from the very conceptual design stage of a reactor system safety and hazards control should be an integral part of the engineering effort. A thorough understanding of what regulatory agencies will require in the way of documentation, pre-analysis, and proposed controlling procedures will save much time and money which might otherwise be spent in redesign of components which fail to pass the various safety reviews which are required, and in rescheduling to accommodate the delays which may be involved. The proper attention to nuclear safety and health physics considerations throughout the evolution of the system will circumvent many problem areas that would otherwise arise and delay the program or restrict the operation of the system when it is ready for test.

10.0 REFERENCES

1. United States Weather Bureau, "Meteorology and Atomic Energy", AECU-3066, July 1955.
2. Church, P.E., "Dilution of Waste Stack Gases in the Atmosphere," Ind.Eng. Chem. 49, (1949) pp. 2753-2756.
3. Sutton, O. G., "The Theoretical Distribution of Airborn Pollution from Factory Chimneys," Quarterly Journal of the Royal Meteorological Society 73, (1947) pp. 426-436.
4. "Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and in Water for Occupational Exposure," Table 1, National Bureau of Standards (U.S.) Handbook, Vol. 69, (1959).
5. Chamberlain, A.C., "Experiments on the Deposition of I^{131} Vapor from an Air Stream", AERE HP/R 1082.
6. Pryor, W.A., "Aircraft Nuclear Propulsion Department Nuclear Safety Guide," GE-ANPD, APEX-715.
7. "Nuclear Safety", GE-ANPD, APEX-921, May 30, 1962 (CRD).

APPENDIX A

OUTLINE FOR A PRELIMINARY HAZARDS SUMMARY REPORT OF A NUCLEAR REACTOR FACILITY*

I. Summary
  - A. General description of facility
  - B. Proposed operation specifications
  - C. Reactor physics
  - D. Reactor control
  - E. Physical structures
  - F. Facility systems
  - G. Accidents analyzed
  - H. Hazards evaluation (Applicant self-evaluation)

II. Introduction

III. Site c. Environmental Analysis
  - A. Geography
  - B. Population density and distribution
  - C. Meteorology
    - 1. Winds
    - 2. Temperature
    - 3. Precipitation
    - 4. Atmosphere stability
    - 5. Atmosphere dust loads
  - D. Hydrology
    - 1. Water flow
    - 2. Ground water uses
    - 3. Tidal effects
    - 4. Restrictions
  - E. Geology
  - F. Topography
  - G. Seismology

IV. Facility Description
  - A. Facility
    - 1. Purpose and scope of operation
    - 2. Functional arrangement of buildings
  - B. Reactor core
    - 1. Nuclear design
      - a. Neutron flux
      - b. Nuclear parameters
      - c. Neutron lifetime
      - d. Fuel configuration
      - e. Fuel cycle
      - f. Metal to coolant ratio
      - g. Required excess reactivity
      - h. Reactivity coefficients
      - i. Reactor kinetics

*Based on a compilation by Captain G. B. Conner, USAF, 1959.

2. Physical characteristics
   a. Fuel assemblies
   b. Support and structural elements
   c. Access holes
3. Heat transfer and fluid flow
   a. Temperatures
   b. Pressures
   c. Velocities
   d. Heat flux
   e. Heat transfer area
   f. Heat capacity
   g. Steam voids
   h. Power density
   i. Hot spot analysis

C. Reactor control
   1. Control rods
      a. Configuration
      b. Reactivity worth
      c. Control rate of change
   2. Rod drives
      a. Drive mechanisms
      b. Linkages
      c. Position indicators
   3. Interlocks
   4. Control action
   5. Neutron source
   6. Testing program

D. Pressure vessels
   1. Design specifications
   2. Penetrations
   3. Testing program

E. Shielding
   1. Thermal
   2. Biological
   3. Blast

F. Instrumentation
   1. Design philosophy
   2. Detailed circuitry
   3. Testing program

G. Cooling Systems
   1. Primary
   2. Auxiliary
   3. Secondary
   4. Emergency and decay heat removed

H. Experimental facilities
   1. Configuration
   2. Purpose
   3. Reactivity effects

I. Auxiliary Systems
   1. Auxiliary control
   2. Ventilation
   3. Fuel storage and handling system
   4. Hot cells
   5. Radioactive waste disposal
      a. Treatment facilities
      b. Storage facilities
      c. Disposal facilities

3. Heat transfer and fluid flow
   a. Temperatures
   b. Pressures
   c. Velocities
   d. Heat flux
   e. Heat transfer area
   f. Heat capacity
   g. Steam voids
   h. Power density
   i. Hot spot analysis

C. Reactor control
   1. Control rods
      a. Configuration
      b. Reactivity worth
      c. Control rate of change
   2. Rod drives
      a. Drive mechanisms
      b. Linkages
      c. Position indicators
   3. Interlocks
   4. Control action
   5. Neutron source
   6. Testing program

D. Pressure vessels
   1. Design specifications
   2. Penetrations
   3. Testing program

E. Shielding
   1. Thermal
   2. Biological
   3. Blast

F. Instrumentation
   1. Design philosophy
   2. Detailed circuitry
   3. Testing program

G. Cooling Systems
   1. Primary
   2. Auxiliary
   3. Secondary
   4. Emergency and decay heat removed

H. Experimental facilities
   1. Configuration
   2. Purpose
   3. Reactivity effects

I. Auxiliary Systems
   1. Auxiliary control
   2. Ventilation
   3. Fuel storage and handling system
   4. Hot cells
   5. Radioactive waste disposal
      a. Treatment facilities
      b. Storage facilities
      c. Disposal facilities

J. Containment
  1. General description
  2. Design criteria and philosophy
  3. Construction
  4. Testing program
K. Component testing program
L. Reactor physics summary

V. Accident Analysis
A. Initiating events
  1. Component malfunction
    a. Loss of coolant
      1. Pressure loss
        a. Pipe rupture or break
        b. Pump failure
        c. Pressure vessel leaks
      2. Valve failure
      3. Leaks
    b. Reactor core failure
      1. Excessive core pressures
      2. Control rod
      3. Fuel element
    c. Instrument failure
    d. Power failure
    e. Structural failure
  2. Reactivity accidents
    a. Operational mishaps
      1. Start-up accidents
      2. Accidental insertions of reactivity
      3. Fuel handling accidents
    b. Inadvertant addition of reactivity
      1. Cold coolant surges
      2. Chemical controls, poisons
      3. Instabilities
      4. Load fluctuations
  3. Experimental accidents
    a. Reactivity effects
    b. Experimental system failure
  4. Accidents due to Acts of God, war, sabotage, etc.
    a. Floods
    b. Severe storms
    c. Earthquakes
    d. War
    e. Sabotage
    f. Aircraft flying over facility
B. Safeguards evaluation
  1. Inherent safety features
    a. Reactivity coefficients
    b. Self-limiting processes
  2. Built-in safeguards
C. Consequences of initiating events
  1. Nuclear excursion
  2. Fuel element meltdown

        3. Chemical reactions (metal-water)
            a. Al-water
            b. Na-water
            c. Al-U-water
            d. Zr-water
            e. Al-Li alloy-water
            f. Zircaloy-water
            g. Hydrogen-oxygen
        4. Credible accidents
        5. Maximum credible accident
            a. Definition
            b. Energy releases
                1. Nuclear excursion
                2. Chemical reactions
                3. Flashing of coolant
                4. Equivalent explosion concept
            c. Physical forces
                1. Pressure vessel
                2. Blast shield
                3. Containment shell
            d. Structural damage

VI. Operational Procedures

VII. Hazards Analysis
    A. Basic criteria
        1. Normal operation
        2. Emergency operations
        3. Radiation standards
            a. Routine exposures
            b. Emergency exposures
        4. Fission product inventory
    B. Routine release of radioactivity
        1. Characteristics of release
            a. Source
            b. Quantity
            c. Type of identity of isotopes
            d. Nature of release (size, volatility, etc.)
        2. Mode of transmission
            a. Meteorological
            b. Hydrological
            c. Direct
        3. Potential receptor dosage
            a. Expected receptor locations
            b. Dosage (integrated)
    C. Emergency releases from credible and maximum credible accidents
        1. Characteristics of release
            a. Source
            b. Quantity
            c. Type of identity of isotopes
            d. Nature of release (size, volatility, etc.)

2. Mode of transmission
   a. Meteorological
      1. Release conditions
      2. Radioactive cloud
         a. Size
         b. Height of rise
      3. Diffusion
      4. Deposition
      5. Rain-out conditions
   b. Hydrological
      1. Surface diffusion
      2. Undergound movement
      3. Sewer systems
   c. Direct radiation
3. Potential receptor hazards
   a. Integrated dosage versus distance
      1. Cloud exposure
      2. Ground deposition
      3. Direct radiation from contained radioactivity
   b. Area contamination, evacuation

D. Emergency procedures
   1. On-site
   2. Off-site

VIII. Evaluation of Facility's Hazards

APPENDIX B

CONTENTS OF TECHNICAL SPECIFICATIONS

(From the Federal Register, April 8, 1961)

Technical specifications for a facility of the type described in 50.21(b) or 50.22 or a testing facility which is also a boiling-water or pressurized-water nuclear reactor, shall include the following items, insofar as they are applicable to the facility concerned. In addition the technical specifications shall include any other items which could have an effect on the safety of operations comparable in magnitude to the effect of the following items.

A. Site

1. Physical location of the reactor plant.
2. Minimum distance to boundary of the exclusion area.
3. Principal activities carried on within the exclusion area.

B. Containment

1. Design pressure and maximum permissible total leakage rate of the containment vessel (including penetrations).
2. Over-all dimensions, materials of construction and free volume of containment barrier.
3. Number, purpose, construction and type of containment vessel penetrations and methods of closure and sealing (including piping, duct-work and access openings).
4. Frequency, pressure, and methods of testing of the containment vessel and penetrations.

C. Primary coolant system

1. General system specifications including:
   (a) Number of loops.
   (b) Flow per loop.
   (c) Minimum loop flow startup time.
   (d) Minimum number of loops operating concurrently.
   (e) Number of pumps per loop.
   (f) Method of coolant circulation and heat removal (normal and auxiliary).
2. Principal reactor vessel design features including
   (a) Pressure rating.
   (b) Material of construction.
   (c) Over all dimensions.
   (d) Types of connections.
   (e) Number of penetraticns.
3. Primary coolant specifications
   (a) Material
   (b) Method of pressurization.
   (c) Maximum permissible activity.
   (d) Number of passes and flow direction through core.
4. Operating variables including
   (a) Minimum core inlet pressure.
   (b) Maximum and minimum core pressure drop.
   (c) Maximum and minimum flow rate.
   (d) Maximum core exit bulk temperature.

5. Principal design features of major components including
   (a) Primary heat exchanger type and rating.
   (b) Maximum primary relief valve settings.
   (c) Minimum capacity of pressure relief system.
   (d) Product specifications and flow rate of purifications system.
   (e) Type sensitivity and flow rate of sampling system.
6. Materials and general configuration of primary system shielding

D. Secondary coolant system

1. Coolant
2. Maximum pressure
3. Maximum temperature
4. Flow rate
5. Minimum condensor vacuum

E. Reactor core

1. Principal core design features including
   (a) Moderator material
   (b) Reflector material and thickness
   (c) Fuel material enrichment and melting or boiling point
   (d) Minimum number of fuel thermocouples
   (e) Clad material and melting or boiling point
   (f) Minimum number of clad thermocouples
   (g) Fuel element nominal dimensions, overall and internal
   (h) Maximum total mass of fuel in the core, by isotope
   (i) Maximum number of fuel elements in the core
   (j) Maximum fuel burnup (MWD)
   (k) Maximum or minimum void coefficent of reactivity and maximum operating void fraction
   (l) Temperature reactivity defect ambient to operating
   (m) Form of burnable poison and method of attachment
   (n) Maximum and minimum reactivity worth of burnable poison
   (o) Type minimum reactivity worth conditions of use and principal design features of auxiliary poison systems.
   (p) Metal to water ratio in core
2. Principal core temperatures and thermal characteristics including
   (a) Maximum thermal power
   (b) Maximum local and average core heat flux (maximum with respect to all variables at rated power
   (c) Minimum burnout safety factor (on heat flux)
   (d) Maximum fuel surface and central temperatures at designated points
   (e) Average power density

F. Control and safety systems

1. Control system design and operating limits including
   (a) Number installed and minimum number of operative control elements and drives, materials of construction and principal design features
   (b) Maximum reactivity worth of automatic control systems and of entire control system hot or cold

(c) Maximum reactivity worth of any individual control system component or gang hot or cold
(d) Minimum shutdown control margin hot or cold
(e) Minimum number of control elements corresponding to minimum shutdown margin
(f) Maximum reactivity addition rate by control elements
(g) Maximum excess reactivity above cold clean critical
(h) Conditions which would automatically cause reactor scram or building closure and activation points for these actions
(i) Type, functions and conditions of use of interlocks
(j) Items which may be bypassed method of bypassing and conditions under which bypassing will be used

2. Safety system design and operating limits including
   (a) Range of period scram use
   (b) Total number and minimum number of operative safety elements and drives, materials of construction and principal design features
   (c) Total reactivity worth of safety elements, hot or cold
   (d) Maximum reactivity worth of any individual safety element or gang, hot or cold
   (e) Maximum reactivity addition rate by safety elements
   (f) Maximum total scram delay time and safety element insertion time
   (g) Minimum number of operative level safety and period safety channels and ranges of use, independence of operations, minimum or maximum redundancy or coincidence etc.
   (h) Minimum worth of safety elements cocked during startup, fuel loading or other core manipulations
3. Characteristics of systems auxiliary to the control and safety systems
   (a) Emergency power supply availability, methods, capacity, uses
   (b) Devices which are activated on automatic building closure

G. Monitoring systems, general design features and specific operating limits including
1. Maximum stack, coolant and building air activity and minimum number and sensitivity of operating monitors for each
2. Maximum radiation level in accessible areas and minimum number and sensitivity of operating monitors
3. Fuel element failure detection equipment sensitivity, localization and sampling interval (if not continuous)

H. Waste disposal systems design and operating features including
1. Equipment for removal of gases or other foreign materials from primary and secondary coolant, moderator reflector or shield; its capactiy and mode of use (continuous or intermittent)
2. Stack height and flow rate
3. Waste holdup capacities, storage and processing methods and maximum activity levels during normal operations, maintenance, refueling etc.

4. Maximum discharge concentrations of liquid and gaseous effluents

I. Emergency cooling system
1. Principal system design features
2. Minimum capacity of emergency heat exchanger
3. Type minimum coolant supply, flow rate, anti power requirement of emergency cooling system
4. Total cooling time made available by emergency cooling system
5. Conditions which would automatically cause emergency coolant initiation, poison injection or other emergency actions

J. Experimental facilities
1. Location, materials of construction, use and principal design features of experimental facilities
2. Maximum total reactivity increase associated with all experiments or experimental facilities by flooding, draining, poison removal, fueled experiment addition or other methods
3. Maximum individual reactivity increase associated with each experiment or experimental facility by flooding, draining, etc.
4. Minimum amount of instrumentation associated with each experiment or experimental facility, including types of sensores, variables sensed, output actions and duplication or coincidence provisions
5. Minimum cooling capacity to each experiment method of cooling and emergency provisions
6. Geometry, pressure resistance and leak rate of experiment containment barriers
7. Significant controls, signals or other mechanisms by which experiments or experimenters (manually or automatically) may affect the reactor control system

K. Administrative and procedural safeguards
1. A provision that the licensee shall have detailed written procedures in effect for all operations which may affect nuclear safety and for emergencies which procedures have been reviewed and approved by responsible officials within the licensee's organization
2. Brief description of the following controls and procedures
   (a) Administrative organization and controls to the extent that these have a potential effect on safety
   (b) Principal operating procedures having a potential effect on safety, including those for initial startup, routine operation, maintenance refueling, conduct and operation of experiments, power/escalation from criticality to full design power and emergencies
   (c) Procedures for and frequency of testing of safety system components, monitors and other equipment having a potential safeguards function
   (d) Procedures for the review within the licensee's organization of proposed modifications in the facility or in operating procedures and of the design and conduct of experiments